IN THE CLAIMS

1. (currently amended) A method for estimating a helium content of a stainless steel core shroud in a boiling water nuclear reactor, the reactor comprising a reactor pressure vessel, a reactor core surrounded by the core shroud positioned in the reactor pressure vessel, the reactor core comprising a plurality of fuel rods and a fuel cycle, said method comprising:

determining a neutron fluence for predetermined areas of the reactor; and

estimating a helium content of the stainless steel shroud at predetermined areas of the reactor using the following equation:

$$C_{He} = 1031 * \left(1 - e^{-b_j * \phi_j}\right)$$

where

C_{He} is the helium concentration as atomic parts per billion of helium in the stainless steel shroud per weight parts per million of boron in the stainless steel shroud;

b_j is a value between about ~~2.50e^{-21} and about 5.00e^{-21}~~ 2.20e^{-21} and about 2.50e^{-21} for a thermal neutron fluence, and between about 3.80e^{-21} and about 5.00e^{-21} for fast neutron fluence;

ϕ_j is fluence expressed as neutrons per square centimeter; and

subscript j denotes thermal fluence or fast fluence.

2. (withdrawn) A method in accordance with Claim 1 wherein determining a neutron fluence comprises:

measuring thermal or fast neutron fluxes in a reactor; and

calculating neutron fluences based on the measured thermal or fast neutron fluxes.

3. (original) A method in accordance with Claim 1 wherein determining a neutron fluence comprises simulating the neutron fluence by using a Monte Carlo radiation transport methodology.

4. (original) A method in accordance with Claim 3 wherein simulating the thermal fluence map by using a Monte Carlo radiation transport methodology comprises:

generating a geometric configuration of a nuclear reactor core and surrounding components;

generating a fuel composition distribution;

calculating three-dimensional nuclide concentrations for the fuel rods and water surrounding the fuel rods using the generated geometric configuration and generated fuel composition distribution;

calculating neutron fluxes using a Monte Carlo radiation transport criticality mode methodology; and

integrating neutron fluxes at predetermined exposure points of the fuel cycle over time to obtain neutron fluences at the predetermined exposure points.

5. (original) A method in accordance with Claim 4 wherein calculating neutron fluxes comprises calculating neutron fluxes using at least one of a Monte Carlo radiation transport criticality mode methodology and a Monte Carlo radiation transport successive criticality mode and fixed source mode methodology.

6. (original) A method in accordance with Claim 4 wherein generating a fuel composition distribution comprises:

calculating node-wise void and exposure distributions of the reactor core; and

calculating exposure and void dependent rod-by-rod nuclide concentrations.

7. (original) A method in accordance with Claim 6 wherein calculating three-dimensional nuclide concentrations for the fuel rods and the water surrounding the fuel rods comprises interpolating the node-wide void and exposure distributions of the reactor core and the exposure and void dependent rod-by-rod nuclide concentrations to generate input for the Monte Carlo radiation transport criticality mode methodology.

8. (withdrawn) A method in accordance with Claim 7 wherein calculating neutron fluxes comprises calculating neutron fluxes using at least one of a Monte Carlo radiation transport criticality mode methodology and a Monte Carlo radiation transport successive criticality mode and fixed source mode methodology at a beginning of a full power phase of a fuel cycle.

9. (withdrawn) A method in accordance with Claim 8 wherein calculating neutron fluxes further comprises calculating neutron fluxes using at least one of a Monte Carlo radiation transport criticality mode methodology and a Monte Carlo radiation transport successive criticality mode and fixed source mode methodology at a middle of the full power phase of the fuel cycle.

10. (original) A method in accordance with Claim 9 wherein calculating neutron fluxes further comprises calculating neutron fluxes using at least one of a Monte Carlo radiation transport criticality mode methodology and a Monte Carlo radiation transport successive criticality mode and fixed source mode methodology at an end of the full power phase of the fuel cycle.

11. (withdrawn) A method in accordance with Claim 4 further comprising calculating neutron fluences at points outside the core wherein calculating neutron fluences at points outside the core comprises:

calculating neutron fluxes using a Monte Carlo radiation transport criticality mode methodology;

saving a surface source based on the core periphery;

calculating neutron fluxes using a Monte Carlo radiation transport fixed-source mode methodology based on the surface source using an adequate number of sampling histories to obtain a predetermined level of convergence.

12. (currently amended) A method for estimating a helium content of a stainless steel core shroud in a boiling water nuclear reactor, the reactor comprising a reactor pressure vessel, a reactor core surrounded by the core shroud positioned in the reactor pressure vessel, the reactor core comprising a plurality of fuel rods <u>and a fuel cycle</u>, said method comprising:

generating a geometric configuration of a nuclear reactor core and surrounding components;

generating a fuel composition distribution;

calculating three-dimensional nuclide concentrations for the fuel rods and water surrounding the fuel rods using the generated geometric configuration and generated fuel composition distribution;

calculating neutron fluxes using a Monte Carlo radiation transport criticality mode methodology;

generating a neutron fluence for predetermined areas of the reactor; and

estimating a helium content of the stainless steel shroud at predetermined areas of the reactor using the following equation:

$$C_{He} = 1031 * \left(1 - e^{-b_j * \phi_j}\right)$$

where

C_{He} is the helium concentration as atomic parts per billion of helium in the stainless steel shroud per weight parts per million of boron in the stainless steel shroud,

b_j is a value between about ~~2.50e-21 and about 5.00e-21~~ 2.20e-21 and about 2.50e-21 for a thermal neutron fluence, and between about 3.80e-21 and about 5.00e-21 for a fast neutron fluence;

ϕ_j is fluence expressed as neutrons per square centimeter; and

subscript j denotes thermal fluence or fast fluence.

13. (original) A method in accordance with Claim 12 wherein calculating neutron fluxes comprises calculating neutron fluxes using at least one of a Monte Carlo radiation transport criticality mode methodology and a Monte Carlo radiation transport successive criticality mode and fixed source mode methodology.

14. (original) A method in accordance with Claim 12 wherein generating a fuel composition distribution comprises:

calculating node-wise void and exposure distributions of the reactor core; and

calculating exposure and void dependent rod-by-rod nuclide concentrations.

15. (original) A method in accordance with Claim 14 wherein calculating three-dimensional nuclide concentrations for the fuel rods and the water surrounding the fuel rods comprises interpolating the node-wide void and exposure distributions of the reactor core and the exposure and void dependent rod-by-rod nuclide concentrations to generate input for the Monte Carlo radiation transport criticality mode methodology.

16. (withdrawn) A method in accordance with Claim 15 wherein calculating neutron fluxes comprises calculating neutron fluxes using at least one of a Monte Carlo radiation transport criticality mode methodology and a Monte Carlo radiation transport successive criticality mode and fixed source mode methodology at a beginning of a full power phase of a fuel cycle.

17. (withdrawn) A method in accordance with Claim 16 wherein calculating neutron fluxes further comprises calculating neutron fluxes using at least one of a Monte Carlo radiation transport criticality mode methodology and a Monte Carlo radiation transport successive criticality mode and fixed source mode methodology at a middle of the full power phase of the fuel cycle.

18. (original) A method in accordance with Claim 17 wherein calculating neutron fluxes further comprises calculating neutron fluxes using at least one of a Monte Carlo radiation transport criticality mode methodology and a Monte Carlo radiation transport successive criticality mode and fixed source mode methodology at an end of the full power phase of the fuel cycle.

19. (original) A method in accordance with Claim 15 further comprises integrating neutron fluxes at predetermined exposure points of the fuel cycle over time to obtain neutron fluences at the predetermined exposure points.

20. (withdrawn) A method in accordance with Claim 19 further comprises calculating neutron fluences at points outside the core wherein calculating neutron fluences at points outside the core comprises:

calculating neutron fluxes using a Monte Carlo radiation transport criticality mode methodology;

saving a surface source based on the core periphery;

calculating neutron fluxes using a Monte Carlo radiation transport fixed-source mode methodology based on the surface source using an adequate number of sampling histories to obtain a predetermined level of convergence.

21. (withdrawn) A system for simulating a helium content of a stainless steel core shroud in a boiling water nuclear reactor, the reactor comprising a reactor pressure vessel, a reactor core surrounded by the core shroud positioned in the reactor pressure vessel, the

reactor core comprising a plurality of fuel rods, said system comprising a computer configured to:

generate a geometric configuration of a nuclear reactor core and surrounding components;

generate a fuel composition distribution;

calculate three-dimensional nuclide concentrations for the fuel rods and water surrounding the fuel rods using the generated geometric configuration and generated fuel composition distributions;

calculate neutron fluxes using a Monte Carlo radiation transport criticality mode methodology;

generate a neutron fluence for predetermined areas of the reactor; and

estimate a helium content of the stainless steel shroud at predetermined areas of the reactor using the following equation:

$$C_{He} = 1031 * \left(1 - e^{-b_j * \phi_j}\right)$$

where

C_{He} is the helium concentration as atomic parts per billion of helium in the stainless steel shroud per weight parts per million of boron in the stainless steel shroud,

b_j is a value between about $2.50e^{-21}$ and about $5.00e^{-21}$;

ϕ_j is fluence expressed as neutrons per square centimeter; and

subscript j denotes thermal fluence or fast fluence.

22. (withdrawn) A system in accordance with Claim 21 wherein said computer is further configured to calculate neutron fluxes using at least one of a Monte Carlo

radiation transport criticality mode methodology and a Monte Carlo radiation transport successive criticality mode and fixed source mode methodology.

23. (withdrawn) A system in accordance with Claim 21 wherein said computer is further configured to:

calculate node-wise void and exposure distributions of the reactor core; and

calculate exposure and void dependent rod-by-rod nuclide concentrations.

24. (withdrawn) A system in accordance with Claim 23 wherein said computer is further configured to interpolate the node-wide void and exposure distributions of the reactor core and the exposure and void dependent rod-by-rod nuclide concentrations to generate input for the Monte Carlo radiation transport criticality mode methodology.

25. (withdrawn) A system in accordance with Claim 24 wherein said computer is further configured to calculate neutron fluxes using at least one of a Monte Carlo radiation transport criticality mode methodology and a Monte Carlo radiation transport successive criticality mode and fixed source mode methodology at a beginning of a full power phase of a fuel cycle.

26. (withdrawn) A system in accordance with Claim 25 wherein said computer is further configured to calculate neutron fluxes using at least one of a Monte Carlo radiation transport criticality mode methodology and a Monte Carlo radiation transport successive criticality mode and fixed source mode methodology at a middle of the full power phase of the fuel cycle.

27. (withdrawn) A system in accordance with Claim 26 wherein said computer is further configured to calculate neutron fluxes using at least one of a Monte Carlo radiation transport criticality mode methodology and a Monte Carlo radiation transport successive criticality mode and fixed source mode methodology at an end of the full power phase of the fuel cycle.

28. (withdrawn) A system in accordance with Claim 23 wherein said computer is further configured to integrate neutron fluxes at predetermined exposure points of the fuel cycle over time to obtain neutron fluences at the predetermined exposure points.

29. (withdrawn) A system in accordance with Claim 28 wherein said computer is further configured to calculate neutron fluences at points outside the core by:

calculating neutron fluxes using a Monte Carlo radiation transport criticality mode methodology;

saving a surface source based on the core periphery;

calculating neutron fluxes using a Monte Carlo radiation transport fixed-source mode methodology based on the surface source using an adequate number of sampling histories to obtain a predetermined level of convergence.